CE...

PERRY NUCLEAR POWER PLANT
10 CENTER ROAD
PERRY, OHIO
(216) 259-3737

Mail Address:
PO BOX 97
PERRY, OHIO 44081

Michael D. Lyster
VICE PRESIDENT - NUCLEAR

February 3, 1992
PY-CEI/OIE-0388L

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
Response to NRC
Confirmatory Action Letter

Dear Sir:

This letter is submitted in response to Confirmatory Action Letter (CAL) RIII-91-016A, dated January 3, 1992 (PY-OIE/CEI 0424L), which discusses commitments regarding the Circulating Water System pipe rupture at Perry Unit No. 1 on December 22, 1991.

Following the December 22, 1991, pipe rupture and subsequent reactor scram, Perry management personnel drafted a recovery plan to determine the extent of damage from the event, evaluate equipment malfunctions and ensure that routine recovery activities were completed. Daily staff meetings were conducted to monitor progress on recovery plan activities.

An Augmented Inspection Team (AIT) was dispatched to the Perry site on December 22, 1992 to respond to the event. On December 24, 1991 CAL-RIII-91-016 was received. This CAL documented commitments regarding actions to be taken prior to making a mode change from cold shutdown. On January 3, 1992, a second CAL, RIII-91-016A, was received closing out the previous CAL dated December 24, 1991, and documenting additional commitments associated with pre-startup and post-startup activities. CAL-RII-91-016A also included an acknowledgment by Region III staff, that designated root cause evaluations and corrective actions specified in the Perry forced outage recovery plan had been completed. After notifying Region III management of completion of the remaining committed pre-startup activities, Perry commenced a plant startup on January 3, 1992.

Provided in Attachment 1 to this response, is a brief description and chronological sequence of events for the December 22, 1991 circulating water pipe rupture. Attachment 2 provides a response to each item discussed in the January 3, 1992, Confirmatory Action Letter.

IE36

Should you have any additional questions regarding this response, please contact Mr. K. P. Donovan, Licensing and Compliance Manager, at (216) 259-3737 extension 5606.

Sincerely,

M D Lyster

Michael D. Lyster

MDL:RWG:ss

Attachments

cc: NRC Project Manager
Region III Administrator
NRC Resident Inspector Office

ATTACHMENT 1

EVENT DESCRIPTION AND
CHRONOLOGICAL SEQUENCE OF EVENTS

EVENT DESCRIPTION SUMMARY AND CHRONOLOGICAL SEQUENCE OF EVENTS

At 0138 hours on December 22, 1991, reactor power was increased from 99 to 100 percent power upon completion of a weekly surveillance test. At 0152 hours, an annunciator was received for low circulating water chamber level. At 0154 the Control Room received reports that the motor and diesel fire pumps had started and that the start-up transformer deluge system had initiated. It was also reported at that time, that a large vapor cloud was seen in the vicinity of the Unit 1 start-up transformer. At 0157, Control Room personnel observed that the cooling tower basin level was rapidly decreasing and that pump amperage and discharge pressure were oscillating considerably for the existing Circulating Water System (N71) configuration. Decreasing vacuum in the "A" auxiliary condenser was also noted.

At 0200 hours, the Control Room Unit Supervisor (US) ordered a decrease in reactor power to 80 percent. This action was taken with the assumption that the "A" auxiliary condenser could be isolated to stop the system leakage. The Control Room personnel thereafter noticed that vacuum was also decreasing in the "B" auxiliary condenser. There were subsequent reports to the Control Room of water in the transformer yard and Turbine Building. Based upon the above considerations, the US directed entrance into Integrated Operating Instruction (IOI) - 8, "Shutdown by Manual Reactor Scram." Reactor core flow was reduced and a manual scram was inserted at 0205.

At approximately 0210 hours, a plant operator reported to the control room that a massive leak existed at the 36 inch circulating water inlet to the heater bay at the 620 foot level (located in the yard area immediately behind the heater bay). As a result, the Unit supervisor ordered the "A" and "B" circulating water pumps secured. Reactor pressure was being controlled by opening the steam bypass valves in accordance with Plant Emergency Instruction (PEI)-B13, "RPV Control." These valves were used until the reactor pressure had decreased to approximately 700 psig. At 0224 hours, the outboard Main Steam Isolation Valves (MSIVs) were closed because of the imminent complete loss of condenser vacuum. The "C" circulating water pump was also secured. Reactor pressure control was then transferred to the Safety Relief Valves (SRVs).

At 0259 the Shift Supervisor declared an Alert due to reports of rising water level in the Intermediate Building, Auxiliary Building and Turbine Building heater bay.

From 0222 to 0657 hours, 50 individual manual SRV cyclings were performed. Reactor Core Isolation Cooling (RCIC) was used to augment the SRV pressure control. As a result of the above actions, reactor pressure was reduced from 674 psig to 128 psig.

Both Residual Heat Removal (RHR) pumps "A" and "B" were operating in the suppression pool cooling mode during the SRV cyclings. The "A" RHR pump was eventually shifted from suppression pool to shutdown cooling mode at 0737 hours to assist in Reactor Pressure Vessel (RPV) cooldown.

The Motor Feed Pump (MFP) and Reactor Core Isolation Cooling (RCIC) were used for reactor level control for most of the transient. While utilizing the SRVs for pressure control, the plant experienced six Level 3 (178 inches above the top of active fuel) actuations and nineteen Level 8 (219 inches above the top of active fuel) actuations due to reactor vessel level shrink and swell as the SRVs were actuated. The Level 3 actuations resulted in scram signal initiations from the Reactor Protection System. No rod motion occurred from these Level 3 actuations since all rods were previously inserted during the manual full scram. The Level 8 actuations resulted in the tripping of the MFP and subsequent failure to restart after its 15th Level 8 trip at 0359 hours. RCIC was again started for level control at 0404 hours.

The initial NRC notification regarding this event was made at 0311 hours to report the Alert declaration. Follow-up notifications were made at approximately 1 hour intervals thereafter. Additional information was transmitted in response to telephonic information request by Region III and NRR personnel. Required notifications to state and local officials were also made in a timely manner. The Alert was terminated at 1151 hours on December 22, 1991.

CHRONOLOGICAL SEQUENCE OF EVENTS
Circulating Water System Rupture
December 22, 1991

0152 - Annunciator received for low Circulating Water chamber level.

0154 - Automatic start of Diesel Fire Pump and Motor Driven Fire Pump; indication of Deluge System initiation on Startup Transformer.

0200 - Low pressure indications on Circulating Water Pump discharge pressure; Cooling Tower Basin low level alarms; major rupture identified on Circulating Water System and cavitation reported. Operators reduce power to 80%.

0205 - Reduced recirculation flow to 52 MLBS/HR and initiated manual Reactor Scram in accordance with IOI-8.

0210 - Plant operator reported leak in 36 inch circulating water inlet piping to Auxiliary Condensers. Secured A and B Circulating Water Pumps.

0224 - Manually closed Outboard MSIVs; established pressure control using Safety Relief Valves. Level was maintained using Motor Feed Pump. Circulating Water Pump C was secured.

0259 - ALERT declared in accordance with Emergency Plan.

0311 - NRC notification made to report Alert declaration.

0400 - After level 8 trip caused by SRV cycling, MFP failed to restart. RCIC used to maintain RPV level.

0737 - Shutdown Cooling established using RHR loop A.

1107 - Entered Cold Shutdown.

1151 - Terminated ALERT; entered Recovery Phase.

ATTACHMENT 2

RESPONSE TO CONFIRMATORY ACTION LETTER
(CAL) RI[illegible]-91-016A

RESPONSE TO CONFIRMATORY ACTION LETTER (CAL) RIII-91-016A

CAL Item 1

Prior to startup, instrument the auxiliary circulating water system inlet flange and the new base plate to measure any movement.

Response to CAL Item 1

This activity was completed on January 2, 1992. The referenced instrumentation was installed under Work Order (WO) 92-00018. As stated in AIT Report No. 50-440/91026 (DRS), this instrument will remain in place until the analysis referenced in CAL Item 3 is completed.

CAL Item 2

Determine quantitative acceptance criteria for movement of the fiberglass to steel flanged portion of the auxiliary circulating water piping prior to startup. Subsequent to plant startup, if you determine that these acceptance criteria have been exceeded, proceed with an orderly shutdown.

Response to CAL Item 2

The acceptance criteria for piping movement is included in Temporary Instruction (TXI)-0131. This instruction includes the maximum readings for the installed instrumentation discussed in CAL Item 1 above and the monitoring frequency for these instruments.

An Operations Standing Instruction, dated 1/3/92, contains the required actions to be taken in the event that the acceptance criteria of TXI-0131 are exceeded. These actions include a verification of failed acceptance criteria prior to initiating a plant shutdown.

CAL Item 3

Within 30 days of making the initial mode change, provide an analysis of the stresses in those portions of the auxiliary circulating water piping system potentially involved with, or affected by, repairs and pipe support modifications.

Response to CAL Item 3

The requested analysis is included in Enclosure 1 to Attachment 2.

CAL Item 4

Within 30 days of making the initial mode change, submit to NRC Region III, a formal report of all significant issues involved in this event including short term and long term recommendations.

Response to CAL Item 4

The requested report is included in Enclosure 2 to Attachment 2.

CAL Item 5

Prior to the end of the refueling outage currently planned for March 1992, make any modifications to piping and pipe supports which are indicated as necessary, if any, as a result of the analysis addressed in Item 3.

Response to CAL Item 5

Any required design changes identified as a result of the referenced analysis will be completed prior to the end of Refueling Outage (RFO) 3.

ATTACHMENT 2

ENCLOSURE 1

AUXILIARY CIRCULATING WATER
PIPING SYSTEM ANALYSIS

AUXILIARY CIRCULATING WATER PIPING SYSTEM ANALYSIS

I. OBJECTIVE

To demonstrate the long-term design adequacy (both piping and supports) for the above ground portion of the N71 Auxiliary Circulating Water Piping System; with sufficient inherent margin to preclude, with a high degree of confidence, future catastrophic piping failures similar to the December 22, 1991 event.

II. BACKGROUND

As part of the root cause evaluations, near-term corrective actions and follow-up activities associated with the December 22, 1991 event, the following were performed and/or concluded:

1. Laboratory analysis of the failed bolts of anchor support 1N71-H0013 (see Addendum A) concluded that nuts on all four (4) baseplate bolts were not tight during system operation prior to the piping failure.

2. The same laboratory analysis concluded that the fiberglass piping catastrophically failed first, with subsequent failure of all four bolts due to extreme overload caused by the water discharge. The primary basis for this conclusion is the severe deformation (bending) present in the failed bolts. Refer to Addendum B for photographs of the failed bolts which have been sectioned.

3. Anchor supports 1N71-H0013 and 1N71-H0021 (inlet and outlet piping, respectively) were redesigned and modifications field implemented during the forced outage via Design Change Package (DCP) #91-0288 to significantly upgrade baseplate anchorage strength and resistance to loosening. Drillco Maxi-Bolts (3/4" diameter) were now used for this application (replacing the previous Hilti Drop-In anchors), which have the following desirable design characteristics:

   - <u>"Ductile" Failure Mode</u>: The bolts are designed per the ductile (i.e., steel fails first) design criteria of ACI-349, Appendix B (Steel Embedments). Design/ultimate loads for such anchors are typically substantially larger and more consistent (less scatter in test data) than for the same size concrete expansion anchor (such as a Hilti Bolt).

II. BACKGROUND (continued)

3. (continued)

- High Initial Bolt Preload: Bolts are installed with high preloads (approximately 80% of yield stress which is about 85,000 psi or 28,000 lbs/bolt). This provides significant resistance to anchor/baseplate loosening.

Refer to Addendum C for an engineering sketch of the redesigned support (1N71-H0013 presented; 1N71-H0021 similar).

4. Consistent with Confirmatory Action Letter CAL-RIII-91-016A, displacement monitoring instruments were installed prior to startup as follows:

- Horizontal displacement (N-S, axial direction of steel piping) of the steel piping flange at the fiberglass interface is monitored via a dial indicator. Called Dial "A".

- Vertical displacement (up and down) of the steel piping flange at the fiberglass interface is monitored via a dial indicator. Called Dial "B".

- Horizontal displacement (N-S and E-W) of the baseplate of "anchor" support 1N71-H0013 (located approximately 8' south of the fiberglass-steel interface) via a "scratch pad" installation.

Refer to Addendum D which provides an engineering sketch of the installed temporary instrumentation and supporting frame.

Refer to Addendum E for plots of Dial Indicator "A" and "B" readings which have been recorded (per TXI-0131) since January 2, 1992 (startup). For reference purposes, the recorded displacements are also plotted against variation in recorded N71 pipe temperature. As can be seen, there is general trending between the two plots. Concerning the baseplate scratch pad for anchor 1N71-H0013, no movement has been recorded to date as anticipated.

5. Consistent with CAL-RIII-91-016A, a displacement "acceptance criteria" for displacement at the steel-to-fiberglass flanged interface was to be determined. The resulting criteria was established as:

Maximum reading (N-S and up-down vertical) = 125 mils

II. BACKGROUND (continued)

5. (continued)

For the technical bases for this acceptance criteria, refer to Addendum F, Section III.A and Appendix I. Note that Appendix I, under ANALYSIS RESULTS, indicates these initial analyses were tentative pending review/verification. This effort has subsequently been completed by Gilbert/Commonwealth, Inc. (G/C), plus the additional analyses of Section III in essence replace the initial evaluations since they are more definitive in scope as discussed below.

The pertinent considerations from the above were factored into the additional piping analyses (see III below) that were performed pursuant to CAL-RIII-91-016A.

III. ADDITIONAL PIPING ANALYSES

Note: Refer to Addendum F for additional details of the analyses. Addendum F is a summary technical report (G/C Report EA-182, "N71 Pipe Rupture Evaluation") from Gilbert/Commonwealth, Inc. who was contracted by CEI to assist in the follow-up technical evaluations.

Additional analyses were performed to achieve the stated objective of Section I above. An overview of the scope of these additional analyses is presented as follows:

1. Expanded Model

The truncated model which was used to develop the displacement acceptance criteria of Section II.5 above was expanded to include the additional N71 steel piping to the auxiliary condensers, as well as the fiberglass piping to the tie-in (underground) with the 12' diameter Fiberglass Reinforced Plastic (FRP) piping. This was done to ensure enveloping of potential influencing factors of any significance. Further, because of the importance of ensuring the functional integrity of the anchor supports (1N71-H0013 and 1N71-H0021), a detailed finite element model of the support was constructed and utilized in the piping analysis. For specific details, refer to Addendum F, Appendix II/Part A.

III. ADDITIONAL PIPING ANALYSES (continued)

2. Hydraulic Loadings

In addition to deadweight, pressure and thermal loads on the piping system, hydraulic loads were also determined. Even though initial judgements were that these loads are relatively insignificant, they were calculated to obtain added confidence that flow transients were not a credible root cause contributor of the piping failure. The hydraulic loadings which were considered included both steady state impulse loads (due to momentum changes at elbows) and flow transients due to pump starts/stops and valve openings/closures). The details are presented in Appendix III of Addendum F.

The final results confirm the initial judgements that hydraulic loadings are very small (i.e., less than 5% of the pressure load). Nevertheless, for completeness, these loads were conservatively applied to the analytical model and combined with the other loadings.

3. Operating ("Design") Case

The inlet and outlet piping analytical models (as described above) were separately executed for maximum pertinent operating conditions of deadweight, pressure, thermal and hydraulic loadings.

4. Displacement ("Target") Case

Further, for added assurance of demonstrating ample system margin and functional integrity, an artificially imposed "displacement" case was also executed. This case forced displacement within the piping inlet model corresponding to movements of 0.125" (vertical) and 0.135" (N-S) at the steel-to-fiberglass flanged interface. These target displacement values were obtained by extrapolating maximum recorded data to date (see Addendum E) to the estimated full range of system thermal conditions.

The displacement case analysis is also intended to more definitively establish instrument monitoring acceptance criteria for both the FRP piping and the anchor support.

IV. ANALYSIS RESULTS

1. Operating ("Design") Case

The analyses for both the inlet and outlet N71 piping demonstrate ample margin for both piping and supports. With regards to the FRP piping, the maximum calculated stress is 2232 psi, resulting in an additional factor of safety of 1.7 against the long-term strength for FRP pipe of 3800 psi. Anchor support loads were well below allowables.

2. Displacement ("Target") Case

Maximum FRP piping stress for this case is calculated to be 1948 psi, or about a 1.9 additional factor of safety compared to the 3800 psi long-term strength.

For the anchor support, the full range of target displacement values were not quite achieved (approximately 85% of target values for the "functional-check" case). However, this is not to imply that anchor support loss-of-function would immediately occur at the target displacement values. The limiting component for the anchor support's strength is the Drillco Maxi-Bolts. As discussed above in Section II.3, Drillco Maxi-Bolts are ductile which means that the full strength of the steel can be developed (i.e., any bolt overload will cause steel yielding and eventual ductile fracture after significant deformation).

Since the piping displacements in question are thermally driven (and thus limited) and the Drillco Maxi-Bolt steel has an elongation capability of approximately 20% at fracture, it follows that minor additional displacement to meet full target values (from 85 to 100%) will not mean functional failure of the support. Further, the analyses to date, have shown relative insensitivity of the FRP piping to displacements of the magnitude of the target criteria. Thus, very slight increase in piping displacements due to Drillco Maxi-Bolt yielding to reach target piping displacement values, would not be of any significance.

It should also be noted that the calculated limiting displacement values for the anchor support envelope the recorded N71 displacements to date, Addendum E, with ample margin (approximately 1.6, minimum).

Refer to Addendum F for more details of the analytical results.

V. CONCLUSIONS

1. Long-term design adequacy of the subject piping, with respect to system functional integrity, has been demonstrated for the current system configuration. This includes demonstrated capacity (with margin) above N71 piping displacements recorded to date.

2. As discussed within Addendum F (Section IV), the Drillco Maxi-Bolt components of the anchor support achieve approximately 60% of the target displacement criteria when conservative standard design allowables are used. Although system functionality has been demonstrated (Item #1 above), in a desire to provide additional conservatism and still further assurance of system integrity, as well as to alleviate concerns over a repeat catastrophic FRP piping failure, an anchor support redesign will be pursued for both inlet and outlet N71 piping. Planned implementation for this design change will be prior to exiting RFO-3.

ATTACHMENT 2

ENCLOSURE 1

ADDENDUM A

(8 PAGES)

CENTERIOR ENERGY

CX-5

MEMORANDUM

TO J.P. Eppich ROOM PYE110 FROM M.S. Bridavsky DATE 1/15/92
PHONE 43718 ROOM ESS
SUBJECT Analysis of Failed Bolts from the Support System of PNPP Circulation Water Piping Supply Line to the Auxiliary Condensers - Final Report

Four bolts and eight nuts were supplied for failure analysis.

The analysis of the bolts and nuts brought us to the conclusion that the anchor type support of the pipe had a lateral movement which caused a displacement of the pipe. The extent of the displacement could not be determined.

The laboratory analysis of the bolts and nuts indicated that the bolts failed as a result of the overload due to pipe failure.

Laboratory Observations

Four failed bolts and eight nuts were submitted for failure analysis. Most of the bolts and some nuts were cut longitudinally by PNPP personnel.

The bolts and nuts submitted were part of the support of the circulating water piping supply line to the auxiliary condenser from the cooling tower. The supply line is a 36" diameter fiberglass pipe connected to the steel pipe approximately 8' from the support.

The fiberglass pipe comes out from the ground and is connected to a horizontal run of steel pipe. The falure occured at the fiberglass elbow.

The four failed bolts had been secured by Hilti Drop-in assemblies embedded in a concrete base. Each bolt had two nuts. A plate was supposed to rest on the lower nuts and be held down by the upper nuts. To this plate another plate was welded. To this second plate a stand made of a pipe was welded and the 36" steel pipe was welded in turn to the post. That structure constituted an anchor support for the pipe which was designed to prevent pipe movement at this point.

The following observations were made:

1. Two bolts (NE and NW corners) failed below the lower nuts.
2. Two bolts (SE and SW corners) failed between the nuts, within the plate thickness.
3. The threads of three out of four bolts were hammered flat in between the nuts (Fig. 1).
4. All four bolts failed in ductile mode with significant plastic bending.

5. No signs of fatigue on the fracture surface were found.
6. Fatigue cracks were found at other locations on the bolts with indications of corrosion assistance (Fig. 2).
7. Observations made on the nuts showed that only one nut of each couple was in contact with the plate.
8. Chemical analysis (Table 1) showed that nuts and bolts were made of carbon steel. Very low silicon and aluminum content indicated that rimmed steel was used. The Hilti Drop-in assembly was made from low carbon rimmed steel with high sulfur content added for machinability.

Discussion and Conclusion

The observations are summarized in Table 2 and Figure 3. The findings indicate several significant facts.

a. The nuts were not tightened down, which allowed the plate to move (see fig. 3).
b. The plate had a significant lateral movement, which was proven by the signs of friction between the plate and nuts and by the hammered threads.
c. The lateral movement of the plate caused fatigue of the bolts and, possibly, cycling loading of the pipe by allowing the steel pipe to move with the plate it was welded to.
d. The metallographic analysis of the bolts confirmed the presence of fatigue cracks filled with iron oxide at locations other than the fracture surface. Analysis of one of the cracks was performed on SEM by mapping the iron and oxygen contents. As it can be seen on Fig. 4, the crack is filled with iron oxide. That confirms the corrosion fatigue origin of the crack and its slow propogation.
e. If the bolts had failed prior to the pipe failure in the fatigue mode, the failure would have been certainly located below the lower nuts with no significant plastic deformation. The severe bending of all four bolts leads us to believe that the bolts failed due to overload but not due to fatigue.

Based on the laboratory analysis we came to the the conclusion that the fiberglass pipe failed first and the water discharge caused the overload and failure of the bolts.

Reviewed by George P. Bobko

Date Jan 21st, 1992

Attachments
MSB/bjw
cc: R.M. Kantorak
R.F. Katora
R.J. Standish
R.J. Tadych

Table 1

Analysis of failed bolts from
the support system of PNPP
circulating water piping supply
line to the auxiliary condensers

Chemical Composition of Bolts and Nuts

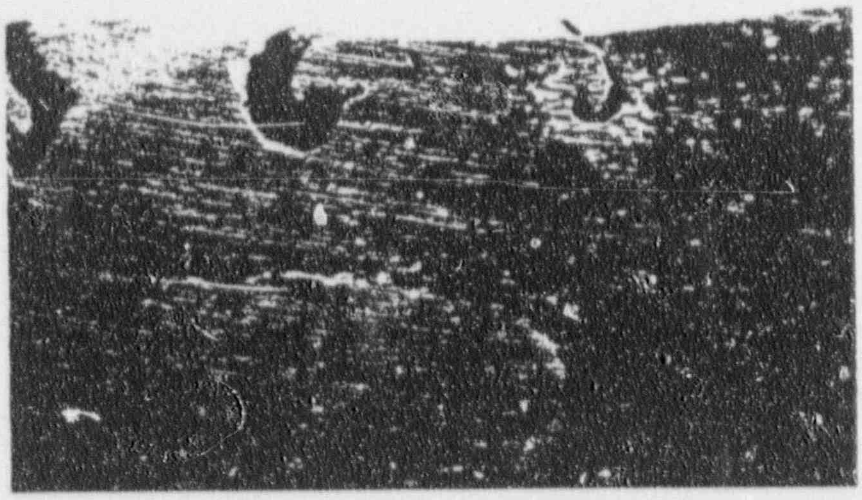
Location (Corner)	Description	Chemical composition, wt %					
		C	Si	Al	Mn	P	S
N.E.	NUT	0.039	0.005	0.007	0.37	0.007	0.040
	BOLT	0.27	0.040	0.007	0.40	0.010	0.046
N.W.	NUT	0.040	0.005	0.009	0.36	0.007	0.35
	BOLT	0.27	0.041	0.007	0.40	0.009	0.039
S.E.	NUT	0.037	0.006	0.006	0.38	0.007	0.043
	BOLT	0.27	0.040	0.006	0.40	0.009	0.035
S.W.	NUT	0.044	0.006	0.008	0.37	0.007	0.009
	BOLT	0.26	0.039	0.007	0.40	0.009	0.037
Hilti Drop-in Assembly	----	0.067	0.009	0.003	1.46	0.078	0.28

Table 2

PNPP Circulation Water Line Failure

Summary of the visual observations.

Corner	Failure Location	Threads Condition	Upper Nut Condition	Lower Nut Condition
SE	Between Upper & Lower Nuts	Hammered on One Side.	Shiny on the Interface w/Plate. Pitted.	Rusted and Pitted. No Signs of Mechanical Friction.
SW	Between Upper & Lower Nuts	Hammering Not Found.	Shiny on the Interface w/Plate. Pitted.	Rusted and Pitted. No Signs of Mechanical Friction
NE	Below Lower Nut, Above the Concrete.	Hammered on Two Sides.	Rusted and Pitted. No Signs of Mechanical Friction.	Shiny on the Interface w/Plate. Pitted.
NW	Below Lower Nut, Above the Concrete.	Hammered on One Side.	Shiny on the Interface w/Plate. Pitted.	Rusted and Pitted. No Signs of Mechanical Friction.

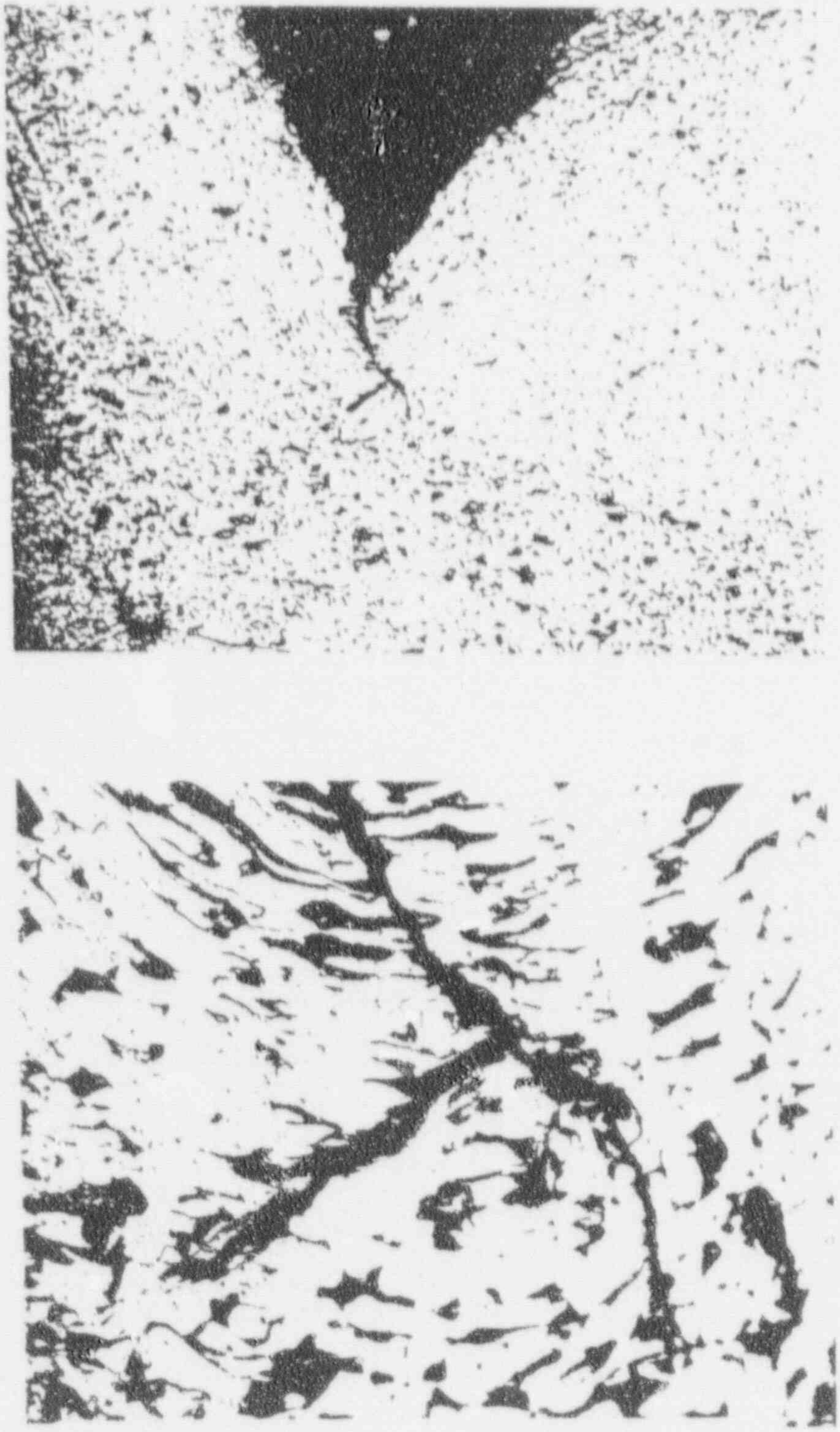


10x

Fig. 1.

PNPP Circulation Water Line Failure

Crossections of the bolt with threads
compressed (hammered).



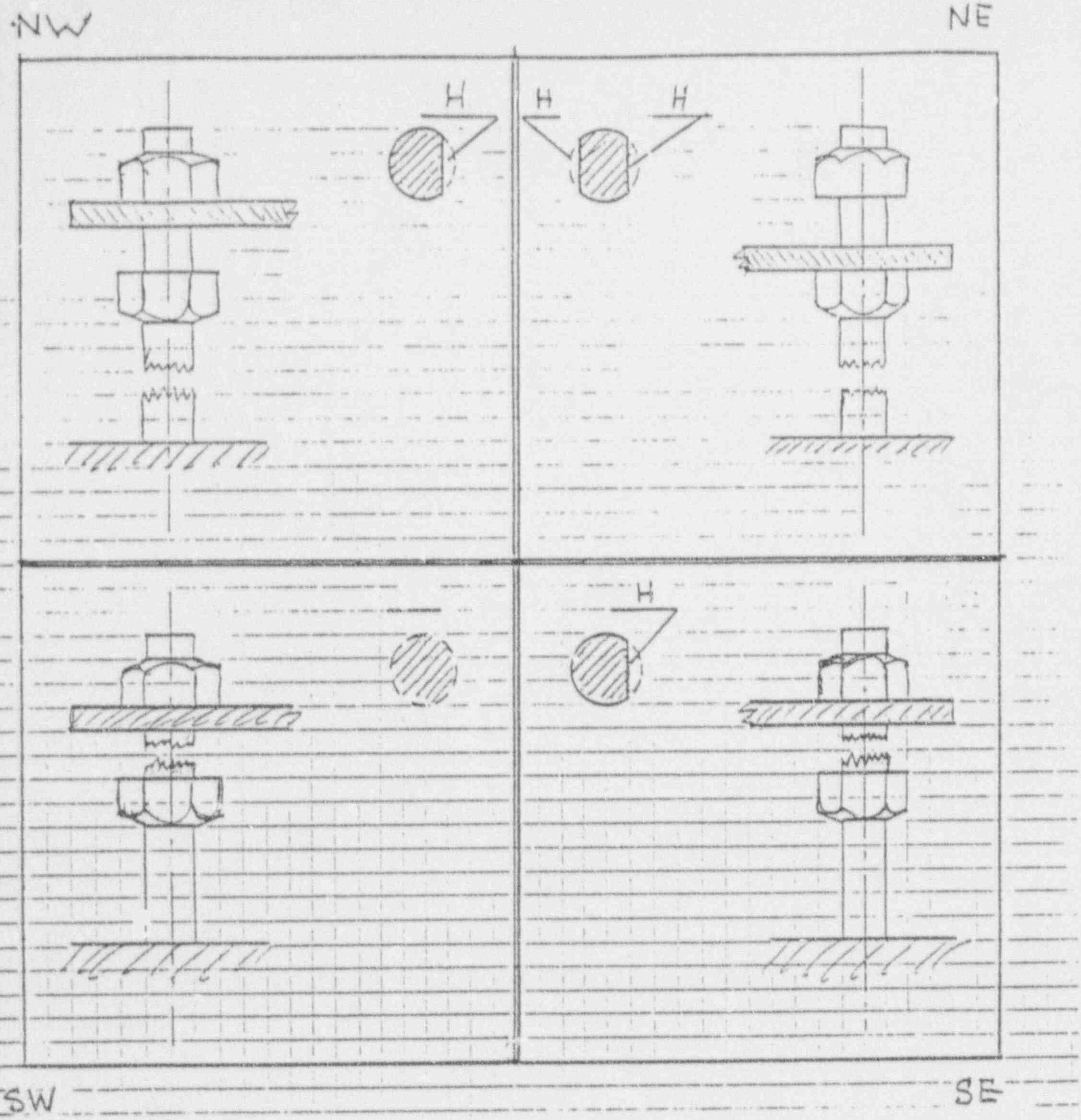
50x

400x

Fig. 2.

PNPP Circulation Water Line Failure

Typical corrosion fatigue crack.



H - THREADS ARE HAMMERED FLAT

Fig. 3.

PNPP Circulation Water Line Failure

Pictographic presentation of the observations summarized in Table 2 and showing the location of the plate in between the nuts.

Fig. 4.

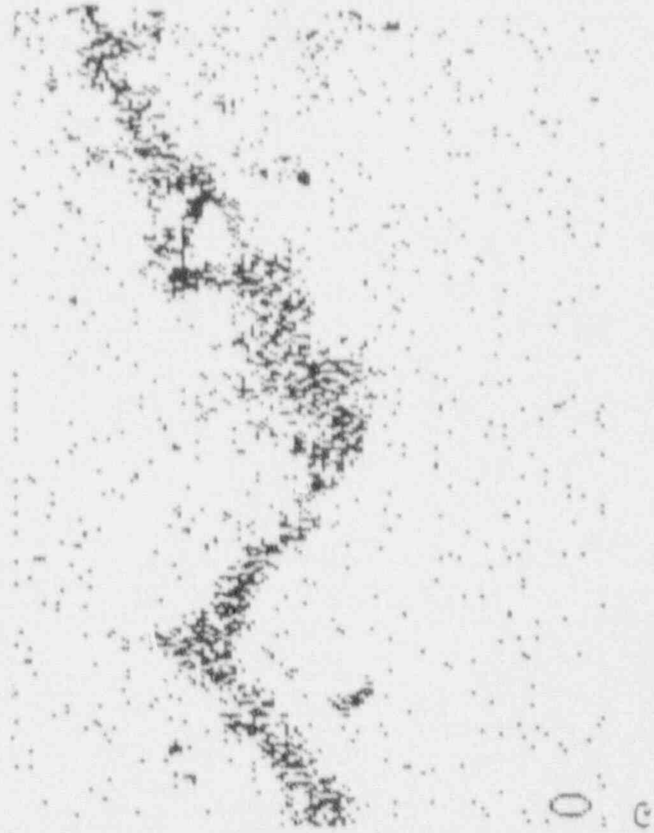
PNPP Circulation Water Line Failure

Stress corrosion crack and chemical elements distribution.

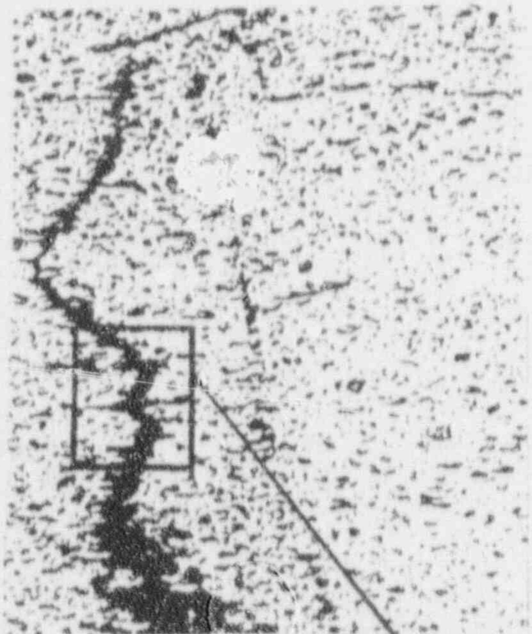
A. Optical photograph of the crack at magnification 50x.

B. SEM photograph of the part of the crack A at 480x.

C&D. EDS map of oxygen and iron.



O C.



A.

OXYGEN

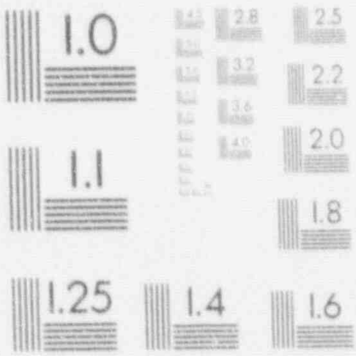
IRON

B.

D

1

IMAGE EVALUATION TEST TARGET (MT-3)



150mm

6"

ATTACHMENT 2

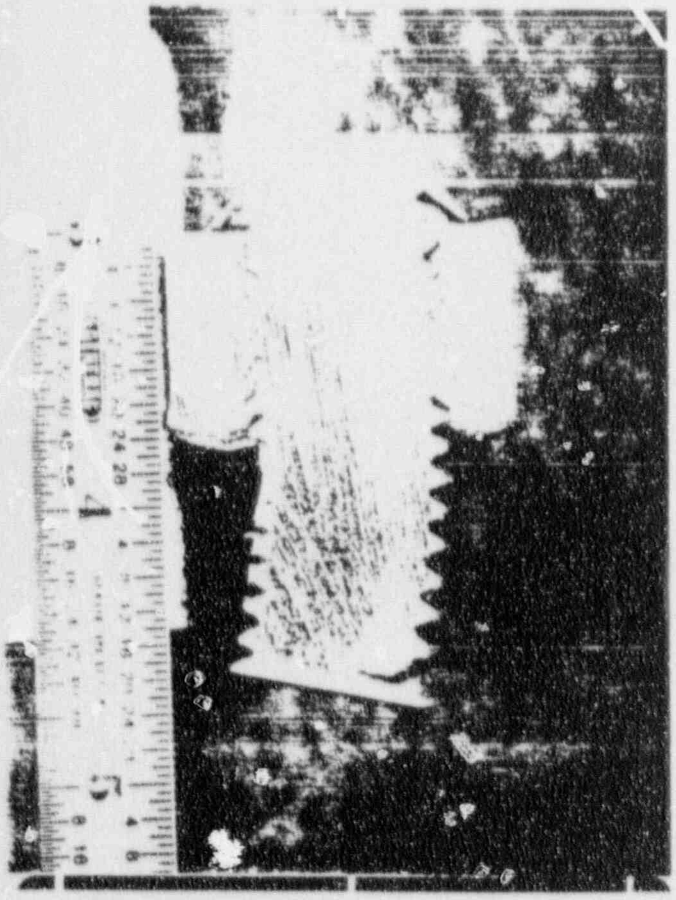
ENCLOSURE 1

ADDENDUM B

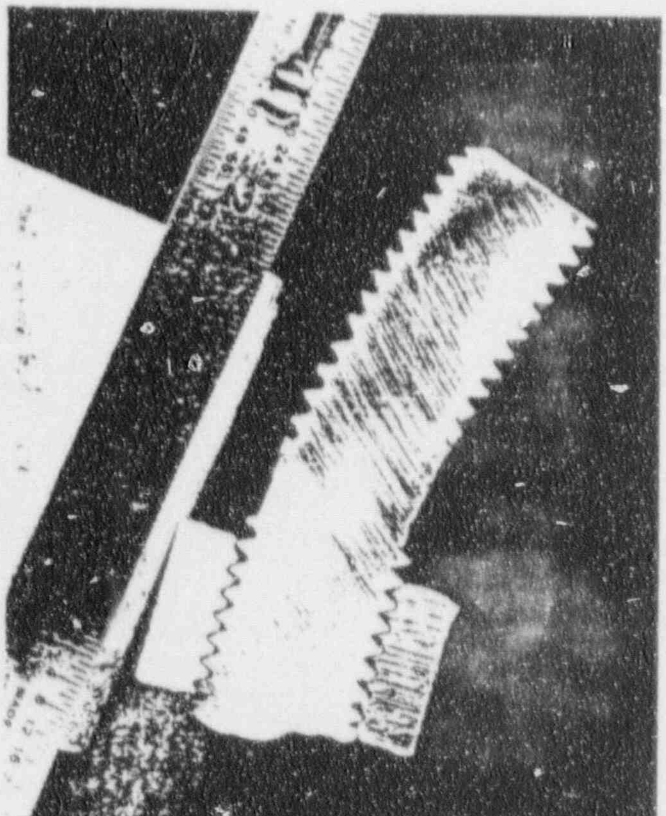
(2 PAGES)

TEST/ANALYSIS REQUEST

NOTE: EXCERPT FROM TAR # 91-514, REV. 1



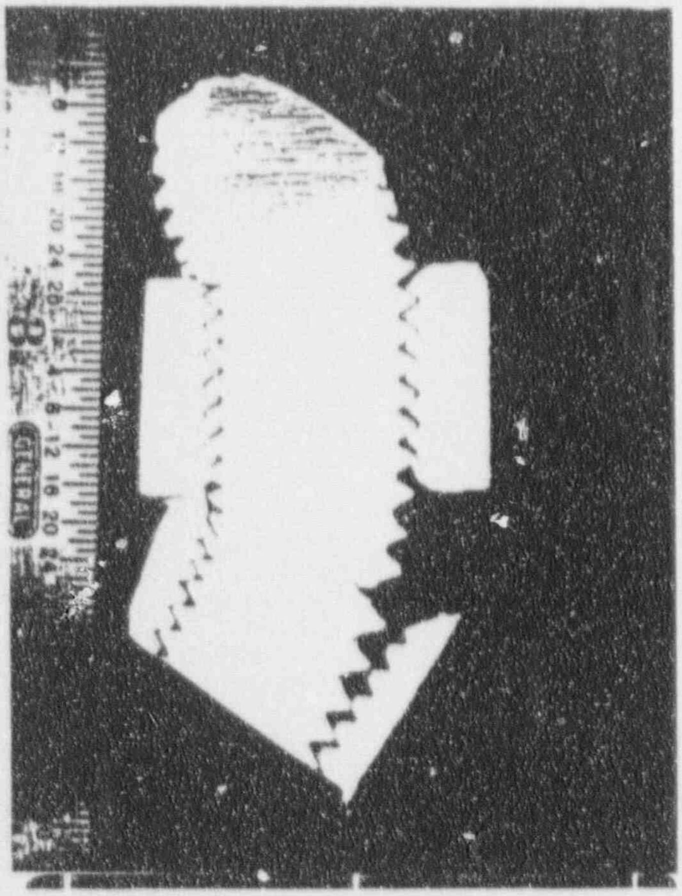
SECTIONED ANCHOR BOLT AND NUT
NW CORNER



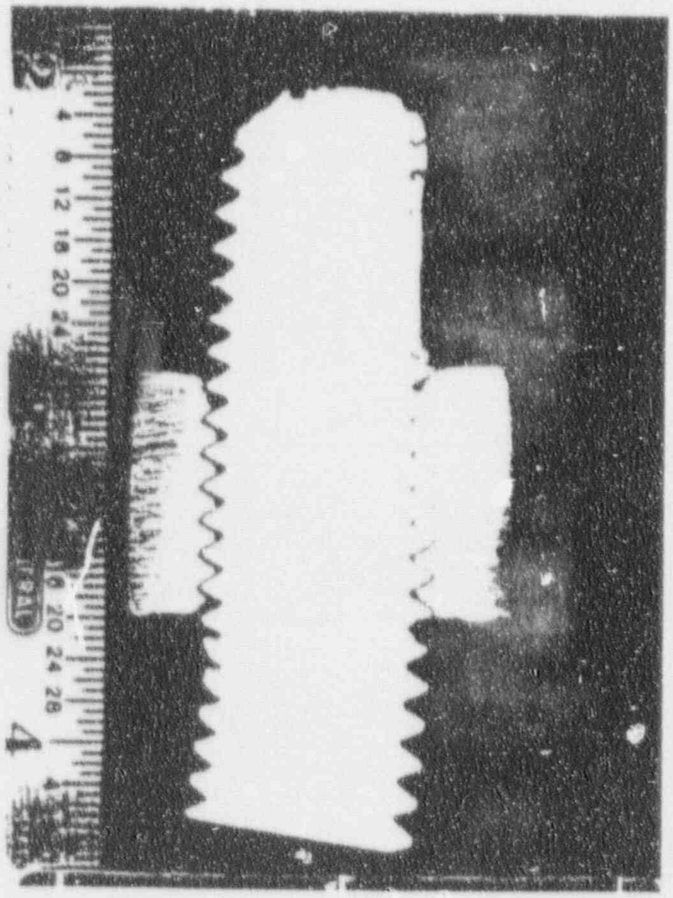
SECTIONED ANCHOR BOLT AND NUT
NE CORNER

TEST/ANALYSIS REQUEST

NOTE: EXCERPT FROM TAR #91-514, REV. 1



SECTIONED ANCHOR BOLT AND NUT
SW CORNER



SECTIONED ANCHOR BOLT AND NUT
SE CORNER

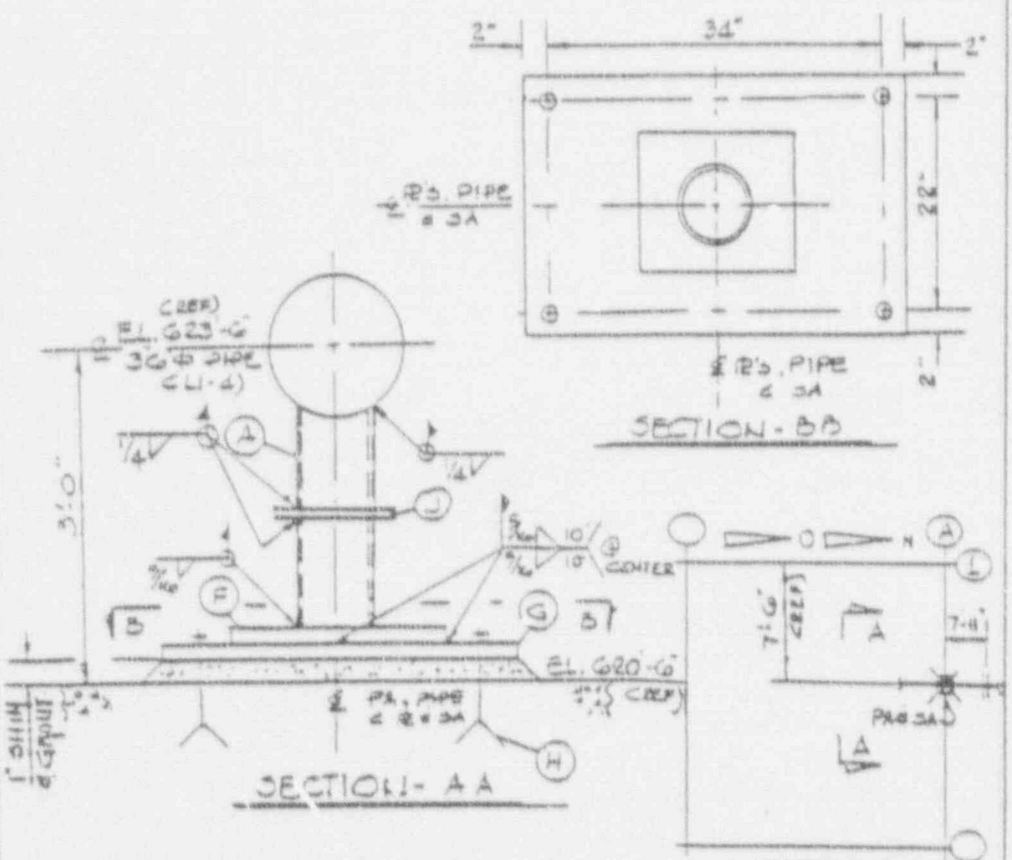
ATTACHMENT 2

ENCLOSURE 1

ADDENDUM C

(1 PAGE)

ITEM	NO. REQ'D	PART NO.	DESCRIPTION	MATERIAL
A	1	-	16" Φ SCH/30 PIPE x 1'-3 7/8" LG. SHAPE TO FIT	
			36" OD PIPE PER SECT. A-A	
B			DELETED	
C			DELETED	
D			DELETED	
E			DELETED	
F	1	-	℞ 1" x 1'-6" x 1'-6" LG	A-36
G	1	-	℞ 1" x 2'-2" x 3'-2" LG W/(4) 7/8" Φ HOLES	A-36
			AS SHOWN ON SECTION - BB	
H	4		3/4" Φ DRILLCO MAXI BOLTS (MB750-13-1/2-9-1/4)	
J	1		℞ 3/4 x 18 x 1'-6" LG	A-36



THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT UNIT 1

4549-92-201-0013

PIPE SUPPORT

MK. 1N71-H0013

TITLE

2 of 2

PAGE

REV

ATTACHMENT 2

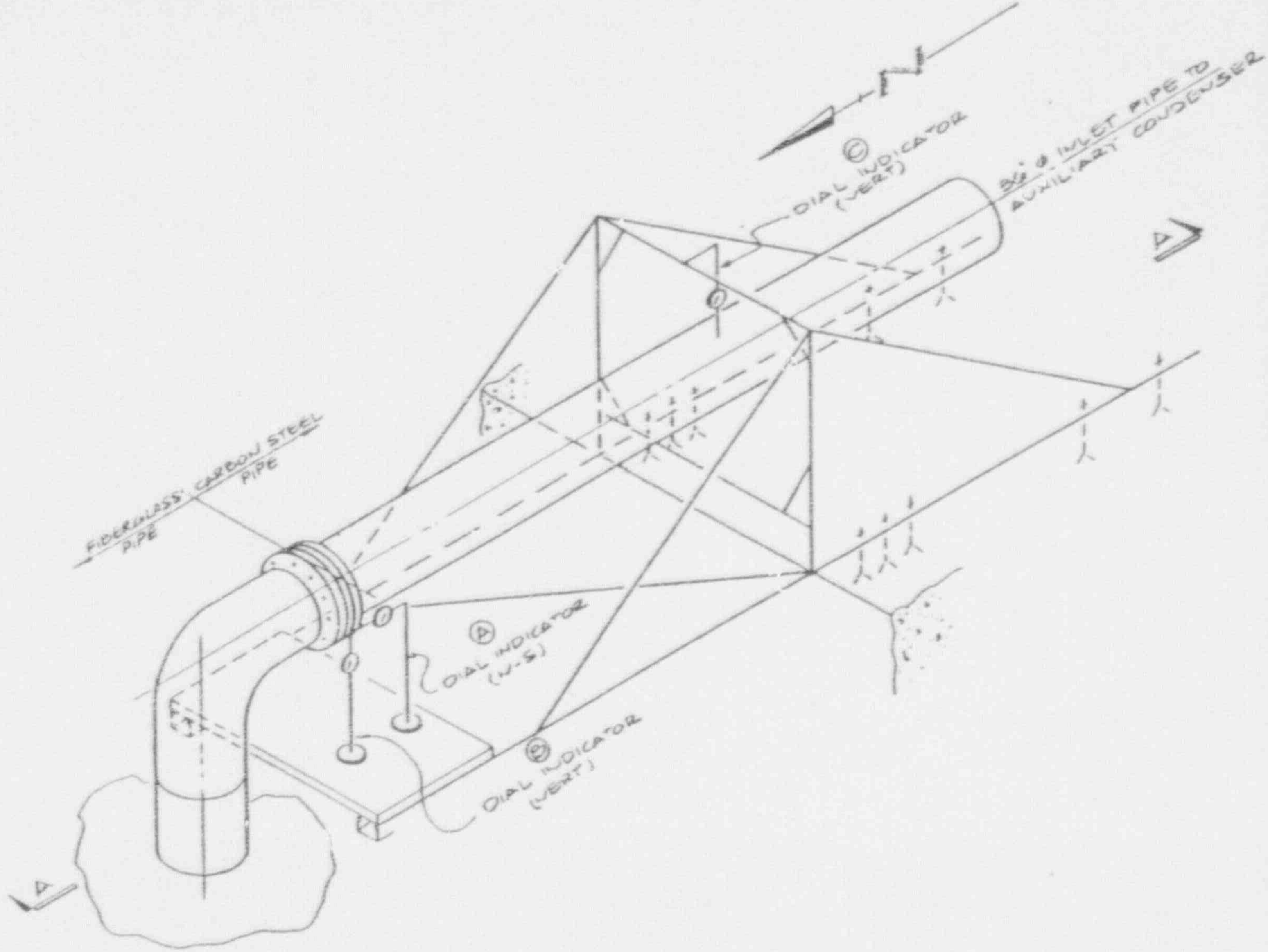
ENCLOSURE 1

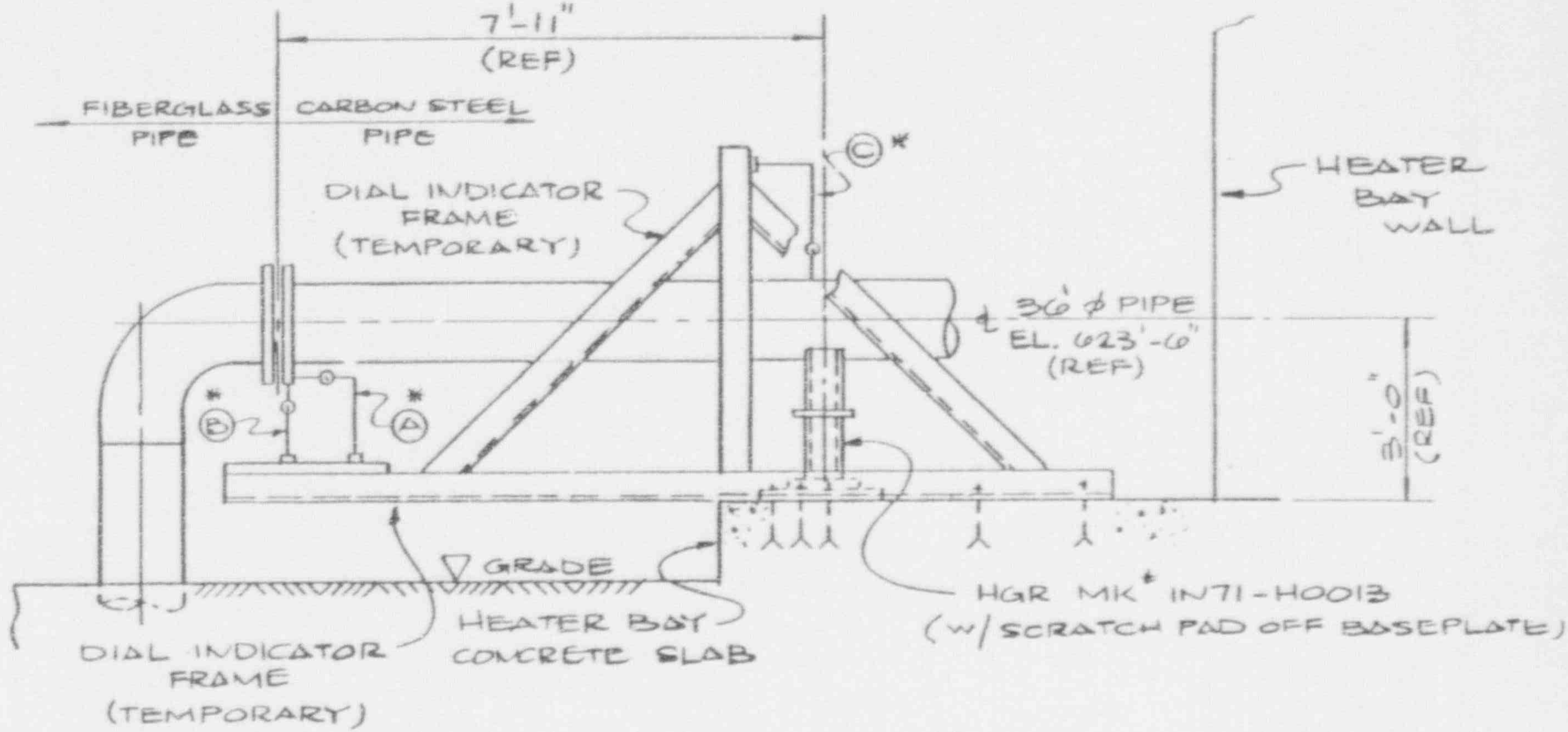
ADDENDUM D

(2 PAGES)

CIRCULATING WATER SYSTEM

TEMPORARY INSTRUMENT LOCATIONS





* DIAL INDICATOR

ELEVATION VIEW

LOOKING EAST

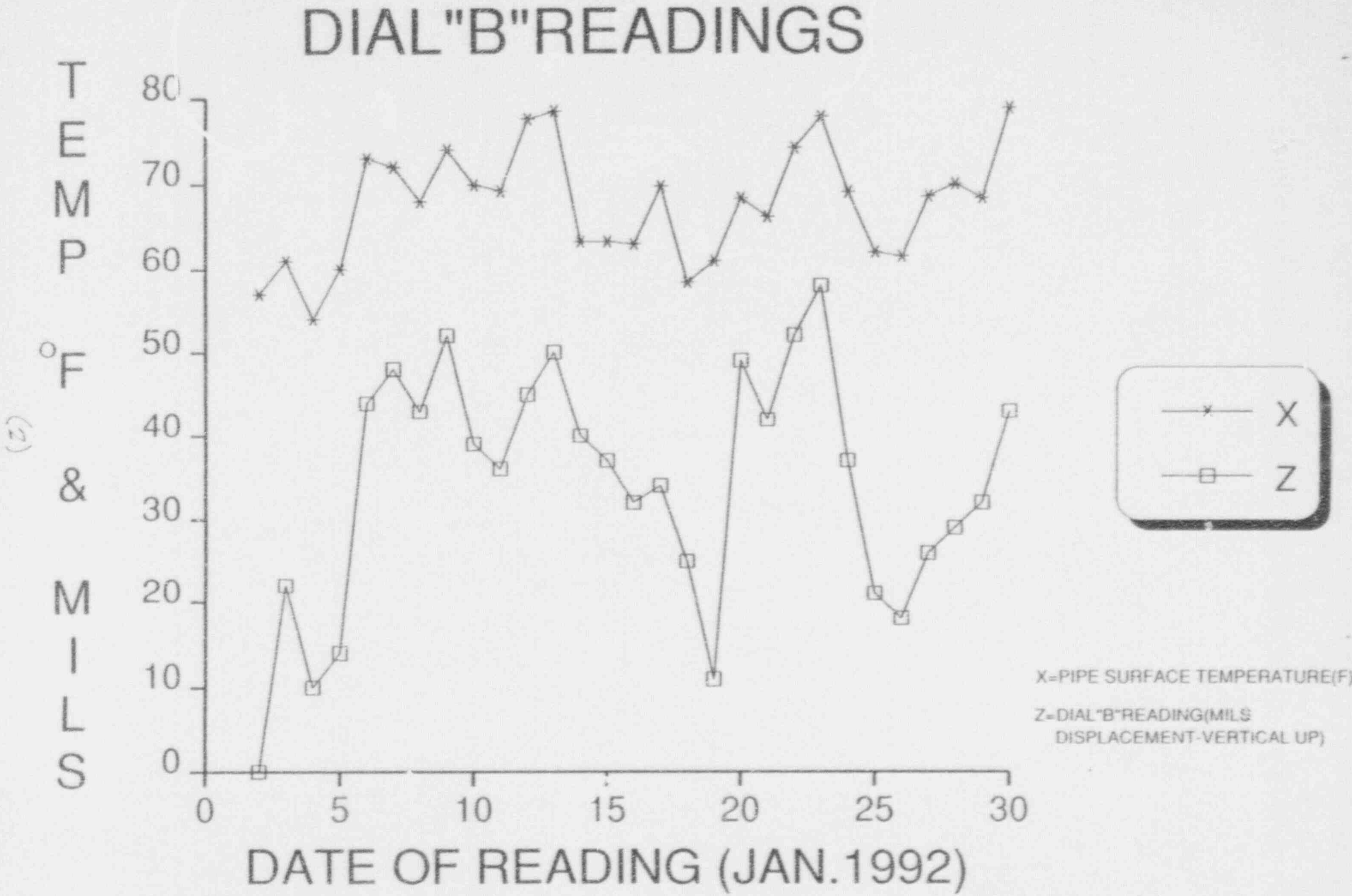
SECTION "A-A"

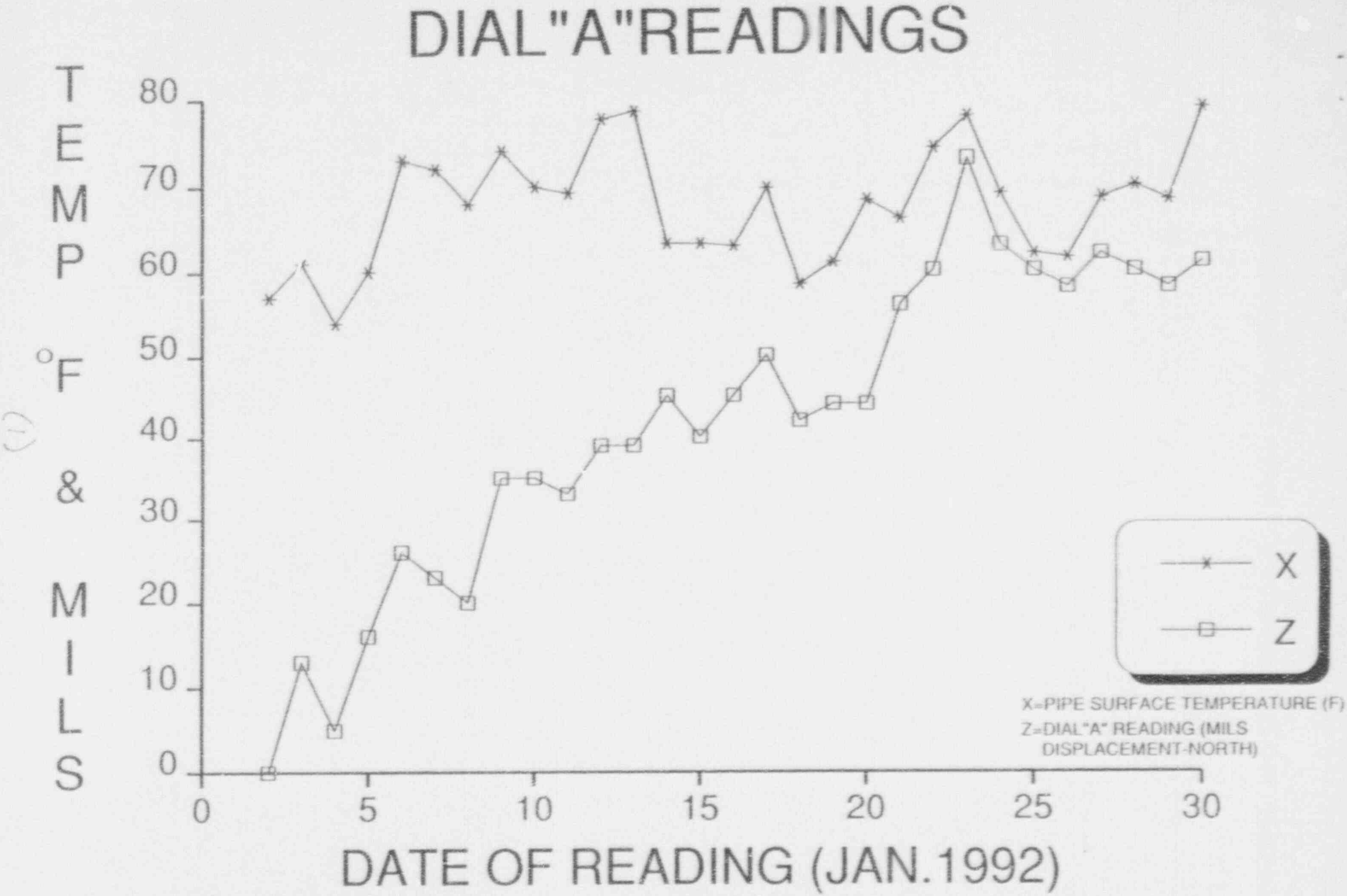
ATTACHMENT 2

ENCLOSU: E 1

ADDENDUM E

(2 PAGES)


DIAL"B"READINGS
TEMP °F & MILS
80
70
60
50
40
30
20
10
0
0
5
10
15
20
25
30
DATE OF READING (JAN.1992)
X
Z
X=PIPE SURFACE TEMPERATURE(F)
Z=DIAL"B"READING(MILS DISPLACEMENT-VERTICAL UP)


DIAL"A"READINGS
TEMP °F & MILS
80
70
60
50
40
30
20
10
0
0
5
10
15
20
25
30
DATE OF READING (JAN.1992)
X
Z
X=PIPE SURFACE TEMPERATURE (F)
Z=DIAL"A" READING (MILS DISPLACEMENT-NORTH)

ATTACHMENT 2

ENCLOSURE 1

ADDENDUM F

(27 PAGES)

<u>NOTE</u>: Pages 5, 12, 13, and 14 of tne attached evaluation were modified on February 2, 1992, and telecopied to CEI prior to submittal of the CAL response.

Gilbert / Commonwealth, Inc. engineers and consultants
P.O. Box 1498 Reading, PA 19603-1498/Telephone 215-775-2600 Cable Gilasoc/Telex 836-431

January 31, 1992

The Cleveland Electric Illuminating Company
Perry Site
P. O. Box 97
Perry, OH 44081

Attn: Mr. J. P. Eppich

Re: Perry Nuclear Power Plant
G/C Report EA-182, Rev. 0
N71 Pipe Rupture Evaluation

Dear Mr. Eppich:

Attached is G/C Report EA-1C2 documenting our evaluation of the N71 circulating water auxiliary condenser inlet and outlet piping and supports. This analysis is in accordance with the task scope defined in Task Authorization #92-0002. We trust that this report will assist you in preparing your final report on this matter.

Please do not hesitate to contact us if you have any questions or need additional information.

Sincerely,

P H Schmitzer

P. H. Schmitzer, P.E.

R J Schmehl

R. J. Schmehl, P.E.

Jan G. Shingler

J. G. Shingler, P.E.
Project Manager

PHS:RJS:JGS:rmb
Attachment

cc: C. R. Angstadt
W. C. Flensburg
PO/DC
J. M. Marrinucci